July 1, 2011

Mr. Anthony L. Patko
Director, Licensing
NAC International, Inc.
3930 East Jones Bridge Road
Norcross, GA 30092

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE THIRD AMENDMENT OF NAC INTERNATIONAL MAGNASTOR® CASK SYSTEM, CERTIFICATE OF COMPLIANCE NO. 1031 (TAC NO.L24470)

Dear Mr. Patko:

By letter dated August 26, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102420569), as supplemented February 4, 2011 (ML11138A224) and February 16, 2011 (ML110480498), NAC International (NAC) submitted an amendment request to the U.S. Nuclear Regulatory Commission (NRC) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 72.244 to amend Certificate of Compliance (CoC) No. 1031. This amendment would revise authorized contents to include pressurized water reactor (PWR):

- damaged fuel contained in damaged fuel cans that are placed in a damaged fuel (DF) basket assembly;
- fuel assembly average burnup up to 70 GWd/MTU;
- fuel assemblies with nonfuel hardware per the expanded definition in this application; and
- fuel assemblies with up to five activated stainless steel fuel replacement rods at a maximum burnup/exposure of 32.5 GWd/MTU.

The first item in the above bulleted list is required to accommodate fuel designated for dry storage at Duke McGuire, Duke Catawba and the Zion Station. The third item in the above bulleted list is required to accommodate hardware nomenclature for fuel designated for dry storage at the Zion Station. The last bulleted item is required to accommodate fuel designated for dry storage at the Zion Station.

This amendment would also revise Paragraph 4.3.1 (i), Appendix A, Technical Specifications (TS), to clarify that the maximum design basis earthquake accelerations of 0.37g in the horizontal direction (without cask sliding) and 0.25g in the vertical direction at the independent spent fuel storage installation (ISFSI) pad top surface do not result in cask tip-over. In a letter dated April 28, 2011, the NRC acknowledged receipt of your amendment request and provided a proposed schedule for our review.

In connection with the staff's review, we need the information identified in the enclosures to this letter. We request that you provide this information by August 29, 2011. Inform us at your earliest convenience, but not later than August 15, 2011, if you are not able to provide the information by that date. To assist us in re-scheduling your review, you should include a new proposed submittal date and the reasons for delay.

Please reference Docket No. 72-1031 and TAC No. L24470 in future correspondence related to this request. The staff is available to clarify these questions, and if necessary to meet and discuss your proposed responses.

If you have any questions regarding this matter, please contact me at (301) 415-3562.

Sincerely,

/RA/

Pamela Longmire, Ph.D., Project Manager
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No.: 72-1031
TAC No.: L24470

Enclosures: 1. RAI (NON-PROPRIETARY)
2. RAI (PROPRIETARY

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NON-PROPRIETARY
Request for Additional Information (RAI)
NAC International
Docket No. 72-1031
Certificate of Compliance No. 1031
MAGNASTOR Amendment No. 3 (TAC No. L24470)

Global Editorial Comment

This amendment request is applicable to pressurized water reactor (PWR) authorized contents. However, revision bars throughout the MAGNASTOR Safety Analysis Report (SAR) submitted in support of this amendment captures text pertaining to the boiling water reactor (BWR). For example, on page 2.2-1, Section 2.2, first paragraph describes spent fuel to be stored. This text captured within the revision bar pertains to BWR fuel assemblies.

Please provide a corrected SAR.

Complete and accurate information must be provided for staff review, per 10 CFR 72.11.

General Information Evaluation

Section 1.1 Terminology

RAI-1.1 Page 1.1-4 For the cited "Nonfuel hardware" to be loaded with fuel assemblies as approved content, identify those with structural functions required for maintaining the as analyzed geometry of a fuel assembly and its ancillaries for safety evaluation of the MAGNASTOR storage system. Clarify if hardware is required to maintain its structural integrity during storage to ensure compatibility for future transportation needs.

It is understood by the staff that some hardware, such as rod cluster control assemblies (RCCAs), may be used to control reactivity of an under-burned fuel assembly in a loaded cask. The 10 CFR 72.236(m) requirements state, "… consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy."

Principal Design Evaluation

RAI-2.1 Section 2.4.2 Confinement Barrier and Systems, Page 2.4-2 Revise the last bullet with the appropriate bold italic lettering annotation, "Using retaining blocks of bolted retaining ring on the transfer cask to ensure that the TSC is not raised out of the transfer cask," by recognizing that a 10 CFR 72.48 change process has been implemented to allow optional use of a retaining ring, in lieu of the retaining blocks, to preclude a load drop accident associated with an inadvertent off-normal lifting of the transfer cask.

SAR Section 3.4.3.3.2 implements a 10 CFR 72.48 change process by performing an evaluation of the retaining ring mounted to the top of the Option 2 MAGNASTOR transfer cask (MTC2). It is not clear if this design change is requested to be within the scope of the present amendment request and is properly annotated.

Complete and accurate information must be provided for staff review, per 10 CFR 72.11.

Structural Evaluation

RAI-3.1 Section 3.2.1 Calculated Maximum Weights and Center of Gravity, Table 3.2.1-1 Revise, with appropriate bold italic lettering annotation, the weight and center of gravity entries that are subject to a 10 CFR 72.48 change process and, therefore, not part of the LAR review.

The table, as annotated, marks every entry as a review candidate and lacks clarity to facilitate staff review by not properly differentiating the items already subject to a 10 CFR 72.48 change process from those covered by the LAR.

Complete and accurate information must be provided for staff review, per 10 CFR 72.11.

RAI-3.2 Section 3.4.3.3.1 Transfer Cask Lift: MTC1, Table 3.4.3-1 With respect to the top-row stress results; explain why the stresses reported for the 9-inch diameter trunnions are identical to those of the initially certified transfer cask equipped with the 9-1/2-inch diameter trunnions.

Difference in stresses is expected for the transfer cask trunnions of different diameters.

Complete and accurate information must be provided for staff review, per 10 CFR 72.11.

Section 3.10.6 Basket Stability Evaluation for Concrete Cask Tip-Over Accident Condition

RAI-3.3 Perform a dynamic transient analysis of the PWR DF basket to demonstrate that it is geometrically stable with acceptable margin for the concrete cask tip-over accident condition. Alternatively, on the basis of similarity consideration, the staff will consider it acceptable for review if a dynamic transient analysis is performed of the finite element model of the initially certified PWR basket except for the modified pin-to-slot connection modeling details. For the evaluation performed, also submit appropriate finite element analysis input and output files for staff review.

The staff expects that a basket geometric stability evaluation similar to those presented in SAR Section 3.10.6 for the initially certified PWR basket assembly be provided for review to meet the 10 CFR 72.122(b) requirements. The analysis input and output files should be submitted per USNRC Spent Fuel Storage and Transportation Division Interim Staff Guidance No. 21 (ISG-21), “Computational Modeling Software,” for staff review.

RAI-3.4 Section 3.10.10.2 Structural Evaluation of the Damaged Fuel Can Spacer, Page 3.10.10-5 Provide a SAR license drawing for the DFC spacer used for limiting the DFC axial movement within the TSC cavity.

Physical attributes of the spacer considered in the structural performance evaluation, including the cross-sectional area and material of construction, should be depicted in a license drawing.

Complete and accurate information must be provided for staff review, per 10 CFR 72.11.

Thermal Evaluation

RAI-4.1 Please refer to the MAGNASTOR calculation packages: Document No. 71160-3127, Rev. 0 and Document No. 71160-3140, Rev. 0. Provide additional information regarding the thermal evaluation of the MAGNASTOR storage system inputs for design basis heat load and ambient conditions. Provide a detailed explanation of how the discretization error was obtained. Also, provide a detailed justification on whether the margins are adequate for the thermal analysis results provided in the SAR. In order to facilitate the NRC review of the application, the analysis results should include an estimate of the numerical uncertainty, grid convergence, and sensitivity of the performed computational fluid dynamics (CFD) analyses. To assist in the technical review, please provide an estimate of the numerical uncertainty and provide a response to the following questions.

a) Has a sensitivity analysis been performed concerning turbulence modeling, boundary conditions, grid independence, and grid convergence?

b) Was grid convergence index (GCI) used to assess uncertainty of the predicted results?

Provide results such as percentage of the calculation discretization error and analysis files used to obtain the GCI. The applicant may consult the following documents for further information on generally accepted CFD best practice guidelines:

(1) Best Practice Guidelines for the use of CFD in Nuclear Reactor Safety Applications, NEA/CSNI/R (2007)5, (ADAMS accession number ML071581053); and

(2) Coleman, H.W. and Stern, F., “Uncertainties and CFD Code Validation,” *Journal of Fluid Engineering*, 119:795-803.

This information is required to show compliance with 10 CFR 72.122(h)(1).

RAI-4.2 Explain how the value listed in the third paragraph, first sentence, under “DCR(L) 71160-SAR-0B/NAC-09-MAG-006,” was changed from 521°F to 550°F on page 3.5-13 of the SAR.

The applicant provided a list of 10 CFR 72.48 changes that constituted the bases for the Amendment 3 application and within these changes, the value for the axial average temperature at the center of the basket is now 550°F, instead of 521°F. An explanation on why this temperature change occurred should be provided.

This information is required to show compliance with 10 CFR 72.122(h)(1).

Confinement Evaluation

RAI-5.1 Provide a detailed justification for how this system meets the requirement of 10 CFR 72.236(j) that requires an inspection and 10 CFR 72.236(l) that requires appropriate tests, or by other means acceptable to the NRC.

The applicant states in SAR 7.1.1 and 10.1.3 that a test envelope will be installed around the transportable storage canister (TSC) enclosing all of the TSC shell confinement welds, and filled with 99.995% pure helium to an acceptable test concentration.

Consistent with the guidance in ANSI 14.5-1997, leak testing of the confinement boundary should encompass welds, joints, and surfaces of the confinement boundary including the base material, the applicant should provide a basis for demonstrating that the materials, forging, fabrication, and testing of the entire confinement boundary construction provide reasonable assurance that leakage through the canister during its entire service life is not credible, without confirmation by helium leak test.

The staff has outlined an acceptable means in Interim Staff Guidance (ISG)-25. Accordingly, the staff would accept helium leak testing the entire confinement boundary including the base material as stipulated in a revised Safety Analysis Report (SAR), Certificate of Compliance (CoC), and its Appendices A (TS) and B (Approved Content and Design Features). This methodology should fully address helium leakage rate testing of the entire confinement boundary welds, including port cover welds, [with the exception of the lid-to-shell weld excluded by ISG-18], and the base material, including the MPC shell, baseplate, lid, port covers, etc. The SAR, CoC, and its Appendices should be consistent with ISG-25 and helium leak testing the entire confinement boundary to be leaktight in accordance with ANSI N14.5-1997.

This information is needed to determine compliance with 10 CFR 72.236(j) and (l).

RAI-5.2 Provide references to address the procedures for helium leakage rate testing in the SAR, as well as the applicable LCO and SR in CoC Appendix A (TS).

The applicant stated in SAR 7.1.1 that during fabrication, the transportable storage canister (TSC) shell and bottom plate welds are volumetrically inspected and the shell assembly in shop helium leakage tested to a leaktight criteria in accordance with ANSI N14.5-1997, using the evacuated envelope test method. Besides the regulations/guidance mentioned in RAI 5-1, the applicant should also provide reference to address the procedures used to perform the helium leakage rate testing of the entire confinement boundary of the storage cask as well as the applicable limiting condition of operation (LCO) and surveillance requirement (SR) in CoC Appendix A (TS), in accordance with ANSI N14.5-1997 and ISG-25, approved on August 18, 2010.

This information is needed to determine compliance with 10 CFR 72.236(j) and (l).

Shielding Evaluation

Observation:

Item	Issue/Question	Correction(s)
Axial Burnup Profile	Figure 5.2.3-3 and Figure 5.2.3-4	Figure 5.3.1.-3 and Figure 5.3.1-4
Axial Burnup Profile	Figure 5.2.3-5	Figure 5.3.1.5
Axial Burnup Profile	Figure 5.2.3-1 and Figure 5.2.3-2	Figure 5.3.1-1 and Figure 5.3.2-2

RAI-6.1 Provide analysis that quantifies the self-shielding provided by the four damaged fuel assemblies.

The SAR states that the damaged fuel model dose rates are less due to the increase in self-shielding from the four damaged fuel assemblies, compensating for the increase in source strength. However, it is not clear to the staff how much self-shielding the damaged fuel cans are providing to the control area boundary dose. The applicant is requested to provide analysis that can quantify the effect of the self-shielding.

This information is needed for the staff to determine compliance with the regulatory requirements of 10 CFR 72.104 and 72.106.

RAI-6.2 Provide technical bases or validation for using the SAS2H code to calculate the gamma and neutron source terms of the fuel with burnup up to 70 GWd/MTU.

The applicant requests to increase the allowable burnup of the fuels to 70 GWd/MTU. However, the SAS2H code of the SCALE 4.4 package used in the source term calculations has not been validated to 70 GWd/MTU. The statements in NUREG/CR-6701 (Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel), "In this demonstration the sensitivity profiles were calculated for a generic LWR fuel assembly having an initial enrichment of 3.0-wt% ^{235}U. The depletion calculations were performed using a nominal power of 35 MW/t and extended to a maximum burnup of 75 GWd/t. The results presented here are only intended as a demonstration to illustrate typical trends in the sensitivity coefficients for several important data parameters with burnup," which the applicant quoted in the SAR, do not appear to be appropriate for use as a basis for code validation. Review of NUREG/CR-6701 indicates that the statements in Appendix A were for the sensitivity study rather than for code validation. NUREG/CR-6701 also shows that the fuel sample with maximum burnup is 73 GWd/MTU was taken from a reconstituted fuel assembly and the value is for peak rod burnup. The corresponding fuel assembly burnup is about 60 GWd/MTU (peak burnup/pellet power peaking factor/rod peaking factor = 73/1.1/1.1 ≈ 60).

The SAS2H has never been validated to 70 GWd/MTU burnup. Statements made in NUREG/CR-6701 clearly indicate that the sample from H. B. Robinson was merely used for illustration of the sensitivity study. In addition, the staff does not find that NUREG/CR-7012 and NUREG/CR-7013 provide an adequate basis for SAS2H for use in shielding applications for high burnup fuel since these documents were written for use in burnup credit applications. The nuclides important for burnup credit analyses are not the same as the nuclides used for shielding analyses.

This information is needed for the staff to determine compliance with the regulatory requirements of 10 CFR 72.104 and 72.106.

RAI-6.3 Provide justification for the selected peaking factor of 1.08 for PWR fuel burnup.

The peaking factor would affect the results of the dose rate calculation. It seems that this peaking factor is not consistent with the study performed in support of NUREG/CR-6801. Based on the data published in NUREG/CR-6801, the peaking factor for PWR fuels seems to be 1.108 rather than 1.08. The applicant is requested to provide justification for the selected peaking factor.

This information is needed for the staff to determine compliance with the regulatory requirements of 10 CFR 72.104, and 72.106.

Criticality Evaluation

RAI-7.1 For License Drawing 71160-601 or 71160-602, include drawings of the screens used to prevent gross fissile material release from a damaged fuel can (DFC) into the transportable storage canister (TSC) cavity. Also, include drawings of the location, number and size of the holes used in each DFC to ensure water drains from the DFC. Provide a justification that the dimensions of the screens are conservative enough to preclude release of gross fissile material from a DFC.

According to the MAGNASTOR, Amendment 3, SAR, a DFC provides a screened container to prevent gross fissile material release into the TSC cavity from failed fuel rod cladding. License Drawings 71160-601 and 71160-602 do not show these screens, and the SAR only states that the “DFC lid and bottom include screened drain holes.” The number of drain holes and the dimensions of the screens are not specified.

This information is needed to confirm compliance with 10 CFR 72.124(a) and 72.236(c).

Materials Evaluation

RAI-8.1 Provide justification(s) for the integrity of the cladding at 68.2 GWd/MTU for the peak temperature the cladding will reach in storage. The justification should account for increased cladding oxide thickness, hydrogen content and cladding stress at the higher burnup.

The staff has indicated in Interim Staff Guidance – 11, Revision 3 that burnups below those levels licensed by the Office of Nuclear Reactor Regulation (NRR) that the cladding will maintain its integrity if the maximum temperature does not exceed 400°C. For burnups higher than the levels licensed by the Office of NRR limit it is incumbent upon the applicant to justify the maximum temperature at which cladding integrity will be maintained.

This information is requested to meet the requirements of 10 CFR 72.122(h)(1) and 72.166.

RAI-8.2 Provide Section 8.2 “Applicable Codes and Standards,” page 8.2-1 of Chapter 8 Materials Evaluation.

This information is required for completeness and accuracy of information and is requested to meet the requirements of 10 CFR 72.11.

Operating Procedures Evaluation

RAI-9.1 Explain how the hydrogen concentration is measured or analyzed to confirm that the 2.4% limit is not exceeded during TSC closure welding.

The applicant provided the procedure in SAR 9.1.1 (step 42) for monitoring the hydrogen generation during loading and closing the transportable storage canister (TSC). The applicant is required to explain how a limit of hydrogen concentration (2.4%) below the closure lid is measured or analyzed to confirm the absence of explosive or combustible gases during TSC closure welding.

This information is needed to determine compliance with 10 CFR 72.122(c) and 72.122(h)(1).

RAI-9.2 Provide more information of drying verification of TSC cavity during loading operation in CoC Appendix A.

The applicant delineated in SAR 9.1.1 (step 60) and Appendix A (TS), Surveillance Requirement (SR) 3.1.1.2 that upon completion of the vacuum drying, the transportable storage canister (TSC) cavity will be evacuated to a pressure of ≤ 3 torr, and then be backfilled with high purity helium. The applicant is required to provide the following information on CoC Appendix A (TS), SR 3.1.1.2:

(1) The test time duration in which the cavity pressure is less than 3 torr, and

(2) The required safety procedures if the pressure in TSC cavity is ≥ 3 torr.

This information is needed to determine compliance with 10 CFR 72.236(j) and (l).

RAI-9.3 Revise the SAR 9.1.1 (step 59) to assure the requirement of the cladding material (for high burnup fuel) in consistency with SRP 1536.

The applicant should revise SAR 9.1.1 (the note of step 59) from "For fuel burnup greater than 45 GWd/MTU, the number of cooling cycles is limited to ten" to "For fuel burnup greater than 45 GWd/MTU, the number of cooling cycles is limited to ten, <u>with cladding temperature variations more than 65°C (149°F)</u>" in consistency with SRP 1536 and previous SERs.

This information is needed to determine compliance with 10 CFR 72.122(h)(1) and 72.128(a)(4).

<u>**Acceptance Tests and Maintenance Program Evaluation**</u>

RAI-10.1 Clarify why there is a revision bar in Section 10.1.1 c).

Complete and accurate information must be provided for staff review, per 10 CFR 72.11.

RAI-10.2 Correct the typo errors in SAR 10.1.3 and ensure consistency of the unit of the leakage rate in SAR and CoC Appendices A and B.

The applicant stated in SAR 10.1.3 "The leakage test is to confirm that the leakage rate for each port cover is ≤ 1×10^{-7} <u>ref.</u> cm^3/s, which corresponds to a helium test leakage rate of ≤ 2×10^{-7} ref. cm^3/s. Following inner port cover welding, a test bell is installed over the top of the port cover and the test bell volume is evacuated to a low pressure by a helium MSLD system. The minimum sensitivity of the Helium MSLD shall be ≤ 1.0×10^{-7} <u>ref.</u> cm^3/s, helium, which is one-half of the allowable leakage criteria for leaktight."

The applicant should remove "ref" (marked by underline) in the statement when referring to the Helium and check the consistency of the units of leakage rate in the entire SAR and Appendices A (TS) and B (Approved Content and Design Features). The unit of (cm3/sec) is used to represent helium leakage rate while the unit of (ref-cm^3/sec) is used to represent the standard air leakage rate.

This information is needed to determine compliance with 10 CFR 72.122(h)(1) and 72.128(a)(4).

Technical Specifications and Operating Controls and Limits Evaluation

Technical Specifications

RAI-13.1 Page B2-4, Table B2-1, Transportable Storage Canister (TSC) with Damaged Fuel (DF) Basket Assembly Verify that the maximum specified PWR fuel assembly weight of 1,680 lbs., including nonfuel hardware, damaged fuel can (DFC) and fuel spacers, is consistent with those considered in evaluating the MAGNASTOR system components.

Section 3.10.10 uses 1,765 lbs., which is heavier than the maximum weight specified in the TS, for the DFC structural evaluation.

Complete and accurate information must be provided for staff review, per 10 CFR 72.11.

RAI-13.2 Revision bars on Page B2-14 through Page B2-21 reference BWR drawings and parameters that are not included in the amendment request. Please clarify.

Complete and accurate information must be provided for staff review, per 10 CFR 72.11.